LHD-TYPE COMPACT HELICAL REACTORS

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Abstract

From the point of view of D-T fusion demonstration reactors, the LHD-type helical reactor designs are studied to clarify design issues for realizing compact reactors, where the major radius R should be as small as possible. The LHD concept is characterized by two advantages; (1) simplified superconducting continuous-coil system and (2) efficient closed helical divertor. Therefore, on the basis of physics and engineering results established in the LHD project, which has already started plasma confinement experiments, two possible approaches on reactor designs are investigated: increasing the toroidal field B_0 in the concept of Force-Free Helical Reactor (FFHR) with a continuous-coil system, and increasing the plasma minor radius a_p in the concept of Modular Heliotron Reactor (MHR) with an efficient closed divertor. Physics and engineering results are presented, including new proposals.

1. INTRODUCTION

Owing to inherently current-less plasma, helical reactors have attractive advantages over tokamaks, such as steady operation and no dangerous current disruption. This work focuses on design issues for reducing the major radius R of the LHD-type helical reactor (l =2, m=10) from the point of view of D-T fusion demonstration reactors on the basis of physics and engineering results established in the LHD project[1]. On March 31 of this year, after the long and tough 8-year construction schedule of LHD, we have successfully produced the first plasma on schedule and started the first cycle of plasma confinement experiments[2].

The LHD concept is characterized by two advantages; (1) simplified superconducting (SC) continuous-coil system and (2) efficient closed helical divertor. Focusing on these advantages, two reactor candidates have been proposed; Force-Free Helical Reactor (FFHR) with a continuous-coil system, and Modular Heliotron Reactor (MHR) with an efficient closed divertor[3]. On this work, to clarify design issues for realizing compact reactors, two possible approaches are investigated: increasing the toroidal field B_0 in FFHR and increasing the plasma minor radius a_p in MHR. In both cases the helical coil-to-plasma distance, δL, for the blanket and shielding is a common constraint, because δL decreases with increasing B_0 or a_p, eventually limiting the reduction of R. Therefore it is necessary to introduce innovative concepts of nuclear blanket systems and coil configurations in both cases.

2. FORCE-FREE HELICAL REACTOR (FFHR)

In order to increase B_0, the force-free-like concept in FFHR-1(l =3, R=20m)[4] is applied again in the LHD-type compact system FFHR-2 as shown in Table.1, which is 2.5 times larger than LHD but a half of FFHR-1 as shown in Fig.1. By reducing the helical pitch parameter, $\gamma=(m/l)(a_c/R)$, from 1.25 in LHD to 1.15, the averaged minor radius hoop force on the helical coils $<f_a>$ normalized by $B_0 I_H$ is reduced to 73% of LHD as shown in Fig.2. At the same time, the clearance δL increases about 5 times of that in LHD as shown in Fig.1. In FFHR-1, the cylindrical supporting structure was proposed to make large maintenance holes at top and bottom regions of the helical coils, where the FEM analyses of supporting structures resulted in the maximum stress below 650 MPa within the allowable stress of 316 LN-type stainless steel[5, 6]. Therefore, in FFHR-2, the cylindrical supporting structure is adopted again as shown in Fig.1 to use a high toroidal field B_0 of 10T with innovative SC materials such as Nb_3Al. It should be pronounced that, as listed in Table 1, the design γ of 1.15 in FFHR-2 is within the experimental range in LHD, where the averaged minor radius a_c of the helical coil can be varied due to the separately controllable 3-layered helical coil.

Table 1 LHD and reactor design parameters.

Parameters	LHD	FFHR-2	MHR
major radius : R	3.9	10	10 m
av. plasma radius : $<a_p>$	< 0.65	1.2	1.7 m
fusion power : P_f	-	1	2.1GW
external heating power : P_{ex}	< 20	70	50 MW
neutron wall loading (MWm^{-2})	-	1.5	2.2
toroidal field on axis : B_0	4	10	6.1 T
average beta : $< \beta >$	> 5	1.8	4.2 %
enhancement factor of τ_{LHD}		2.5	2
plasma density($10^{20} m^{-3}$): $n_e(0)$ 3.8		1.	3.2
plasma temperature : T(0)	> 10	22	13 keV
number of pole : l	2	2	2
toroidal pitch number : m	10	10	10
pitch parameter : γ	1.25 (1.12~1.377)	1.15	1.25
coil modulation : α	+ 0.1	+ 0.1	+ - 0.3
coil gap : Δ_{gap}	-	-	8°
av. helical coil radius : $<a_c>$	0.975	2.30	2.5 m
coil to plasma clearance : δL	0.16	0.7~1.3	0.4~1.3 m
coil current(MA/coil) : I_H	7.8	50	30
coil current density(A/mm^2) : J		(53)	25 30
max. field on coils : B_{max}	(9.2)	13	14 T
stored magnetic energy(GJ)	1.64	147	73

The ignition access in the H-mode operation regime has been analyzed using the time dependent zero-dimensional power balance equation under the simplest control algorithm with the external heating power P_{EXT} and the fueling rate S_{DT} controlled by the fusion power P_f [7]. In this analysis the H-mode indicator defined by $M_{HL}=P_{h,net}V_0/P_{H,th}>1$ is introduced, where the net heating power density $P_{h,net}[MW/m^3]=P_{EXT}/V_0+P_\alpha-(P_b+P_s)$ with the plasma volume V_0, and the H-mode power threshold $P_{H,th}/S_0[MW/m^2]=A_{HL}nB_0[10^{20}m^{-3}T]$. The coefficient A_{HL} is 0.024 at the onset of the heating phase, according to the experimental results in W7-AS, and is assumed to have a hysteresis with smaller value of 0.012 during the ignited operation. The density limit indicator $M_{DL}=n_C(0)/n(0)>1$ is also introduced, where $n(0)_C[m^{-3}]\propto(P_{h,net}V_0B_0/a_p{}^2/R)^{1/2}$ in the LHD scaling. Fig.3 shows that the self-ignition is possible at P_f of 1 GW and a low $<\beta>$ of 1.8% with the τ_{LHD} enhancement factor h_H of 2.5.

A localized blanket concept is newly proposed as illustrated in Fig.1. In the coil-to-plasma space, there is installed the nuclear shielding of 50cm in thickness and the tritium breeder blanket of 30cm[6] at only the outer side. In the divertor region, there is located the breeder blanket, because this region has no limit for radial build and an efficiently thick blanket works as a strong absorber of neutrons. This localized blanket concept is under optimization with ^6Li enrichment to obtain the tritium breeding ratio TBR >1.1 as well as nuclear shielding by using the 3 dimensional Monte Carlo code MCNP-4B with ENDF/B-VI. As a self-cooling tritium breeder the molten-salt Flibe, $LiF\text{-}BeF_2$, has been adopted from safety considerations : low tritium inventory, low reactivity with air and water,

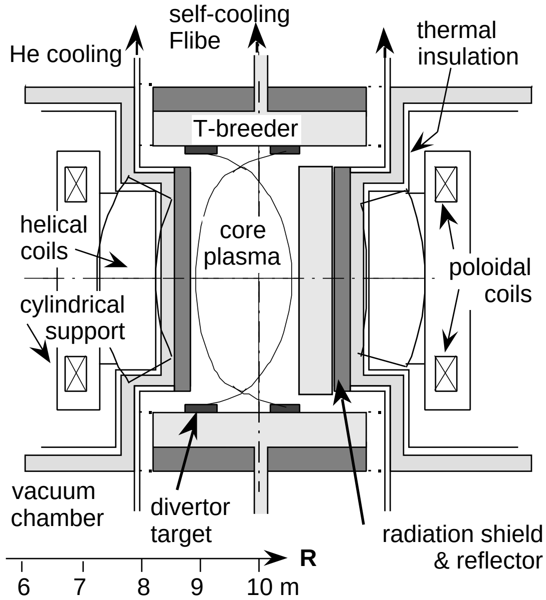

Fig. 1 Schematic illustration of the localized blanket design in FFHR-2.

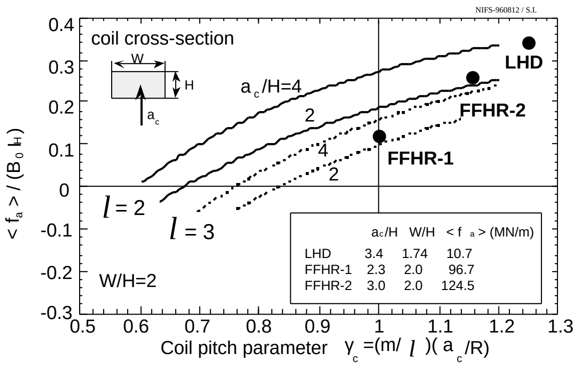

Fig. 2 The averaged minor radius hoop force on helical coils $<f_a>$ normalized by B_0I_H.

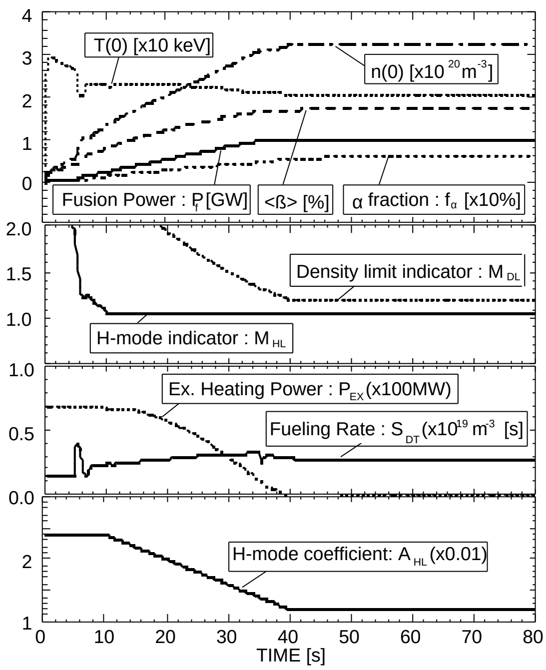

Fig. 3 Ignition access in FFHR-2 .

low pressure operation, and low MHD resistance compatible with high magnetic field. Corrosion of the structural material is mitigated by controlling tritium chemical form to T_2 through contact of Be with Flibe, where Be is used for neutron multiplier. Replaceability of the breeder blankets is also well improved together with divertor targets cooled with Flibe. He gas, which is used to sweep out permeated tritium in double tubes in the Flibe loop, is also used for cooling the nuclear shielding, which is not necessarily needed to be replaced during the lifetime of the reactor, because of the neutron wall loading as low as $1.5 MW/m^2$.

Low activation ferritic steel JLF-1(Fe-9Cr-2W) is the first candidate for blanket structural materials. For higher temperature operation over 550°C, V-alloy (V-4Cr-4Ti) is an alternative candidate because of its low induced activation, high heat flux capability and low irradiation induced swelling. However, the compatibility with Flibe is not well known and needs to be studied.

3. MODULAR HELIOTRON REACTOR (MHR)

MHR, which has the modular coils sectored by toroidal field period (toroidal pitch number, m), has a well-defined and efficient closed helical divertor configuration compatible with modularity [3]. It is the special advantage on maintenance that MHR consists of some identical coils (~10) because it leads to low cost for preparing the standby coils. The coil system of the modular heliotron without one-turn poloidal field coils is constructed based on the conventional heliotron by combining the sectored helical field coils with the sectored returning poloidal coils. Here, the connection current feeders are arranged to avoid the destruction of the divertor layer and to keep a large space for the divertor chamber. It should be noted that the outside-plus/inside-minus modulated windings, $\alpha_{in}<0$ and $\alpha_{out}>0$[8], are important from viewpoint of the good magnetic surface, clean divertor, high MHD equilibrium limit and tolerable neoclassical ripple transport. For the modular system it is difficult to increase the magnetic field because of its complicated structure in the coil system. However, MHR has the property that , as the coil gap, Δ_{gap}, between the modular coils increases, the plasma aspect ratio decreases. It is an approach for the compact system to increase the coil gap. We have carried out the physics optimization of MHR based on MHD equilibrium and stability, neoclassical transport and particle confinement.

Good magnetic configurations are found to be produced by adopting optimum coil modulation parameter as a function of coil gap, Δ_{gap}. Equilibrium beta limits are determined by the criteria for large outward magnetic axis shift from the outermost magnetic surface center (beyond 0.7 of normalized plasma minor radius). From Fig.4 it is obtained that the optimal coil modulation parameter for large MHD equilibrium β limit and large plasma volume are almost same with ones for good branching-off property of the divertor separatorix layers, so that $\alpha_{in}= -\alpha_{out}= -0.038\Delta_{gap}$(degree) is optimal. The MHR with the optimal coil

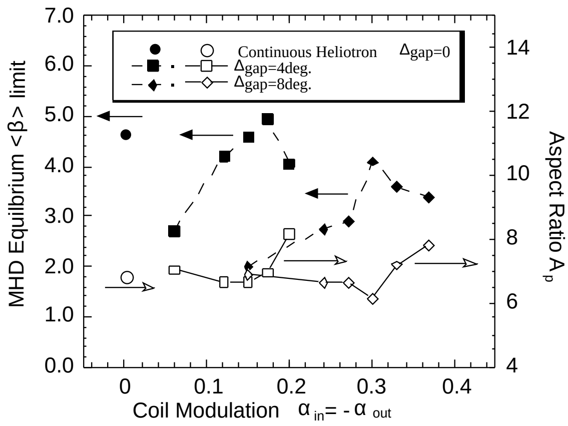

Fig.4 Coil modulation dependence of plasma aspect ratio and MHD equilibrium β limit for modular heliotrons with gap between sectored coils, Δ_{gap}=4° and 8° for m=10.

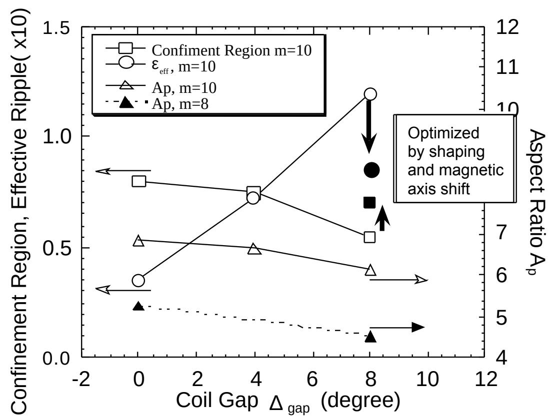

Fig.5 Coil gap dependence of confinement parameters and plasma aspect ratio.

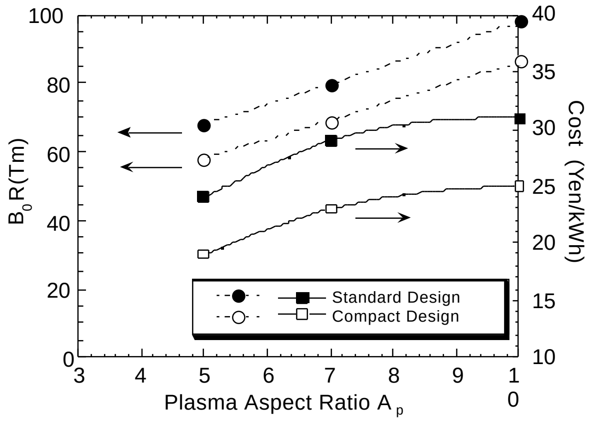

Fig.6 Plasma aspect ratio dependence of major radius, field strength and construction cost.

modulation parameter has almost equal equilibrium performance to conventional heliotron. In the case of less coil modulation, the magnetic shear is not strong, the central rotational transform is small and the magnetic surface becomes horizontally elongated. Then Shafranov shift is extremely large. As concern stability analysis, we apply the Mercier analysis. For $<\beta> \geq 2\%$ in the conventional heliotron, $D_I>0.2$, which corresponds to existence of unstable global ideal interchange mode[9].In the MHR with the optimal coil modulation parameter with $\Delta_{gap}=8°$, $D_I<0.2$ for $<\beta> >1\%$. The stability property is compatible to that of conventional heliotron.

Figure 5 shows coil gap dependence of confinement fraction, effective helical ripple and plasma aspect ratio for MHR system with the optimal coil modulation. The confinement fraction and effective helical ripple amplitude are estimated by minimum-B contour and neoclassical ripple transport model[8]. As coil gap increases, neoclassical transport increases and confinement fraction decreases. By optimizing the magnetic surface shape and magnetic axis position, so that magnetic surface is vertically elongated and magnetic axis shifts torus-inward, confinement properties of MHR are improved as shown in Fig.5. According to transport analysis, effective helical ripple should be less than 5%[3]. It is future subject that the neoclassical ripple transport is reduced further.

Figure 6 shows the dependence of the construction cost and field strength, major radius of MHR on plasma aspect ratio, which is estimated using similar model as Ref.[3]. Open and close symbols correspond to compact design ($\delta L>0.4$m) and standard design ($\delta L>1.0$m), respectively. Here, δL denotes the coil to plasma clearance, which is the key issue of the design of the compact reactor[3]. We adopt compact design to MHR. We show the typical design candidate with 10 modular coil system in Table 1. Here we select the R=10m, $<a_p>=1.7$m, $B_0=6.1$T, magnetic axis torus inward shift and vertical elongated plasma shape and $\Delta_{gap}=8°$(1.4m). It should be noted that MHR coil size is 1.0x1.0m^2 applying a coil current density 30A/mm^2. It has the plasma aspect ratio 6, effective helical ripple 8.7% at $r/<a_p>=2/3$ and $<\beta>$ limit 4.2% and the estimated cost of electricity is 21 yen/kWh.

We can proposal further compact MHR design by reducing m number of the coil system. In Fig.5, the plasma aspect ratio for MHR with 8 modular coils, m=8, is also shown with closed triangles. It is noted that in the case of $\Delta_{gap}=8°$, MHR with m=8 has the plasma aspect ratio 4.5. Based on database for MHR with $A_p=5$, 7, 10, we can estimate some reactor design parameter for MHR with m=8. The roughly estimated design parameters are the following as toroidal field on axis 6.0T, major radius 8.9m and coil to plasma clearance 0.37-1.5m. For a design with m=8 it costs 18 yen/kWh. The physical optimization is future subject for m=8 MHR.

4. SUMMARY

Two possible approaches for LHD-type compact helical reactor designs are investigated, and innovative concepts on nuclear blanket systems and magnetic field configurations are proposed.
(1) The high magnetic-field design of FFHR-2 is proposed by applying force-free-like continuous coil system with the helical pitch parameter of 1.15, which is within the experimental range in LHD. A localized blanket concept is newly proposed to improve replacability of blanket units, where the self-cooling tritium breeder of molten-solt Flibe is adopted from the safety considerations. Optimization of blanket design is in progress using the 3-dimensional nuclear calculation code.

(2) The compact MHR design with large coil gap is proposed. Large coil gap leads to low plasma aspect ratio, but to degradation of confinement performance, which is improved by optimizing magnetic surface shape and magnetic axis position. The possibility of more compact system is also proposed by reducing the toroidal field period from 10 to 8.

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